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Mandly, Charles Robert

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GAMMA RADIATION MEASUREMENTS IN THE UTR-10 SHIELD TANK FACILITY

by

Charles Robert Handly

A Thesis Submitted to the
Graduate Faculty in Partial Fulfillment of
The Requirements for the Degree of
MASTER OF SCIENCE

Major Subject: Nuclear Engineering

Approved:

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I. INTRODUCTION

The designing and building of adequate radiation shielding is one of the major problems that faces the nuclear industry today. The complete inability of the human body to sense a radiation hazard and the lack of sufficient information to determine a precise safe allowable dose rate, have made many of our shield designs extremely conservative. This has resulted in massive radiation shields for most reactors. Portable land units for isolated areas, surface and subsurface ship propulsion units, and the applications of nuclear power in space have pointed out the necessity for the development of less massive shielding.

Much of the shielding research of the past has been accomplished with collimated, monoenergetic radiation sources. This type of study certainly is important in the development of theories and in the study of the energy dependence of shielding effectiveness. However, when one examines the extremely complex radiation spectrum that is emitted by a nuclear reactor, it is not difficult to realize the over-simplification of such studies in shielding design work.

Iowa State University's UTR-10 reactor is equipped with a shield tank facility. This facility provides a strong source of gammas and thermal neutrons that are representative of radiation from a thermal reactor. Its instrumentation includes detectors to measure gamma and neutron radiation.

The purpose of this investigation was to examine the shield tank facility and to determine its effectiveness as a gamma testing installation. Specifically, it was desired to determine the gamma radiation level as a function of position in the tank and reactor power. In addition it was

desired to examine the gamma spectrum for variation with position and reactor power.

II. LITERATURE REVIEW

A search of the literature revealed there are three major shield tank facilities currently in operation, namely the Lid Tank Shielding Facility, the Bulk Shielding Reactor, and the shielding area at the Battelle Research Reactor. The first two facilities are located at Oak Ridge National Laboratory and the third installation is at the Battelle Memorial Institute. Of these three installations, only the Lid Facility uses a scintillation detector for gamma radiation measurements.

Cady (1) describes the Lid Tank Facility and its instrumentation. An anthracene crystal was used with a scintillation detector to measure the gamma level in the tank. The current from the photocathode was measured with a semiautomatic electrometer, calibrated to read the dose rate directly in mr/hr. A standardized radium source was used for this calibration. Price (6) discusses dosimetry applications of scintillation detectors.

Morgan (5) gives a detailed description of the Battelle shielding area. In his report, all of the instrumentation is described, including the fission chamber used for gamma dosimetry.

Casper (2) compares centerline gamma and neutron flux measurements in the Bulk Shielding Reactor with those predicted by three available computer programs. Plots of the results and details of the programs are given in his report.

The three installations described in the literature vary markedly from the UTR-10 facility in size, geometry, existance of fission plates, and the distance from the source. On this basis, it was not expected that dose rates or gamma spectra should correspond to those considered in this investigation.

Raffety (7) is the only previous investigator to do experimental work in the Iowa State University shield tank facility. His work was limited to thermal neutrons, and did not consider the effects of gamma radiation.

Data are available in the literature for the gamma spectrum from thermal fission of U^{235}. Maienschein (4) gives a good coverage of such spectral measurements. However, these data are not beneficial for direct comparison with the gamma spectrum in the shield tank, since it first must pass through 3½ ft of graphite before reaching the tank.

III. DISCUSSION OF THE PROBLEM

It would be highly desirable to give the gamma radiation levels directly in units of mr/hr. This could be accomplished by measuring the current from a scintillation detector or an ionization chamber. At this time, such instrumentation does not exist at this facility. As an alternative approach, it was thought that perhaps the scintillation detector-scaler combination could be calibrated against a standard Co^{60} source to scale cpm to mr/hr for the gammas existing in the tank. To make such an approximation, it is necessary to determine a representative energy value for the gammas in the tank. As mentioned previously, the literature gives data for the energy of gammas from thermal fission of U^{235}. Unfortunately, the gammas in the tank have passed through the core tank, the graphite thermal duct, and the aluminum-plate window before reaching this facility, making the values in the literature inapplicable to this problem. Because of the energy dependence of the mass absorption coefficient, it was hoped that the values of this coefficient might indicate a representative value for the gamma energy. Experimentally determined mass absorption coefficients measured in this investigation indicates gamma energies that range from 2½ to 3 Mev. Using the standardized Co^{60} source submerged in the tank, a mass absorption coefficient of 0.0229 cm2/gr was obtained. From the data listed in Grodstein (3), this would correspond to a mass absorption coefficient of a collimated 0.70 Mev monoenergetic x-ray. Failure of this energy value to approach the known gamma energies for Co^{60}, made any calibration based on the previous assumptions highly questionable. In addition, it would have to be assumed that the scintillation detector had the same counting efficiency for the gammas in

the tank as for the Co^{60} source. On this basis, it was decided to give all gamma intensities in terms of cpm or its normalized counterpart.

Background counts were made for each of the 35 predesignated positions in the tank, prior to each run. A background spectrum was recorded prior to each run in which spectral data were to be collected. Over the two months during which the experimental data were being collected, the background was found to be constant, providing the reactor power was maintained at or below 10 watts and the reactor core was permitted to cool a minimum of 12 hours between runs. During the period of this investigation one 10-W run was made. Approximately two weeks were required to allow the background to approach its initial value. Since the purpose of this investigation was to examine the experimental facility, the background values were not subtracted from the measured counting rates. Certainly the background gammas from the long-lived fission fragments will be present in all experimental work, and therefore must be considered. At power levels of 1.0 watts and higher, these background values become insignificant, even if considered.

IV. EQUIPMENT AND PROCEDURE

The UTR-10 reactor at Iowa State University has a 5 ft by 6 ft by 11½ ft high shield tank facility located on the north end of the reactor. The radiation passes from the reactor core to the tank via the 2½ ft by 2½ ft by 3½ ft long shield tank duct. The duct is composed of 4-in. square blocks of graphite, which are protected from the water by an aluminum-plate window at the water-duct interface. The base of the window is 9-in. above the bottom of the tank. An instrument bridge spans the top of the tank and holds the drive mechanism for a 4-in. diameter aluminum probe. The bracket on the bottom of the probe provides for attachment of radiation detectors. This arrangement provides for movement in all three directions, although no provisions are made for rotation of the probe. Steel tapes attached to the tracks of the bridge and on the probe provide a coordinate system for identification of positions in the tank. The top of the tank and the instrument bridge are shown in Figure 1. A cross section of the tank and reactor core are shown in Figure 2, from which the relative position of the tank and the reactor core may be seen.

Thallium activated sodium iodide crystals were used with scintillation detectors to measure the gamma radiation in the tank. A Radiation Counter Laboratories scintillation detector, model 11000, was used to measure the gammas at the predesignated positions in the tank. The detector was contained in plexiglass encasement, which can be seen in Figure 3. The electrical connections to the detector were made through a 15 ft flexible tube. Since the 11000 detector can not be used with a radiation analyzer, all spectral data were collected with a Nuclear Chicago scintillation detector,

Figure 1. UTR-10 shield tank facility with instrument bridge

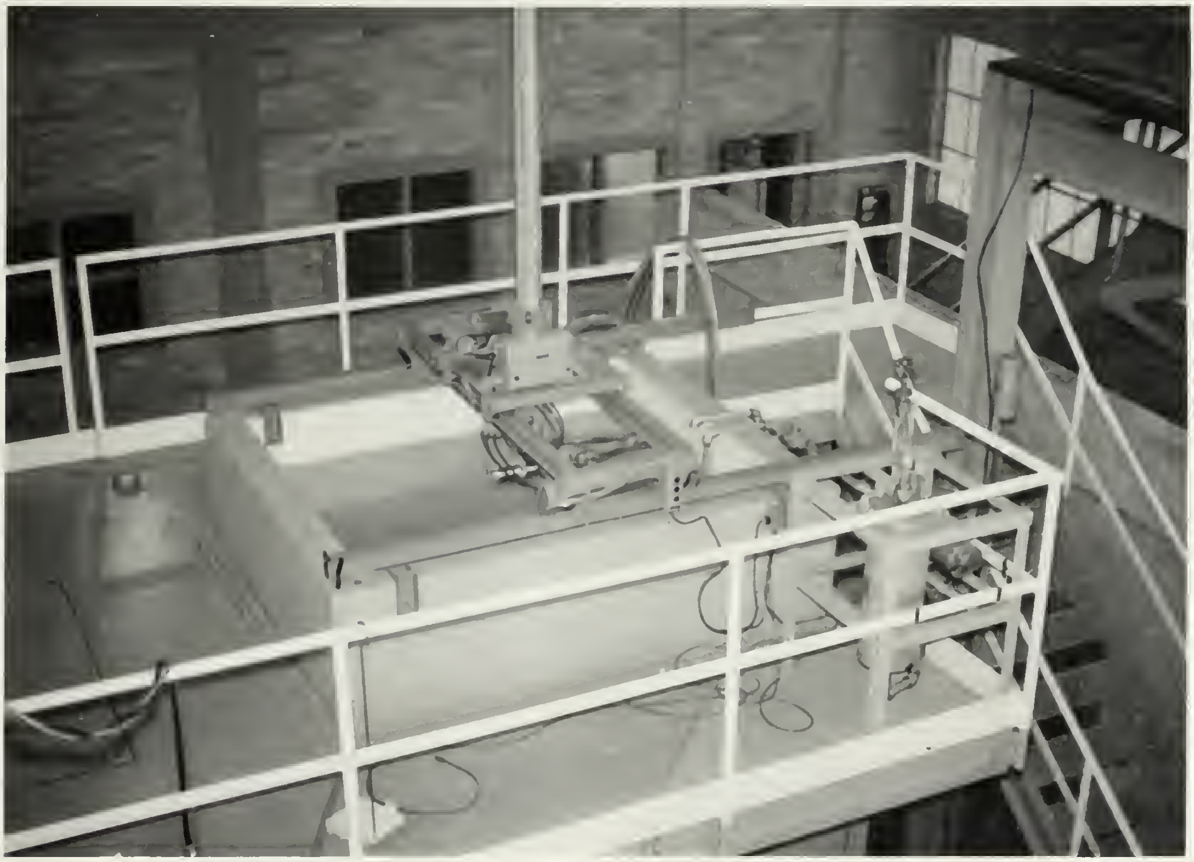


Figure 2. Cross section of UTR-10 reactor and shield tank facility

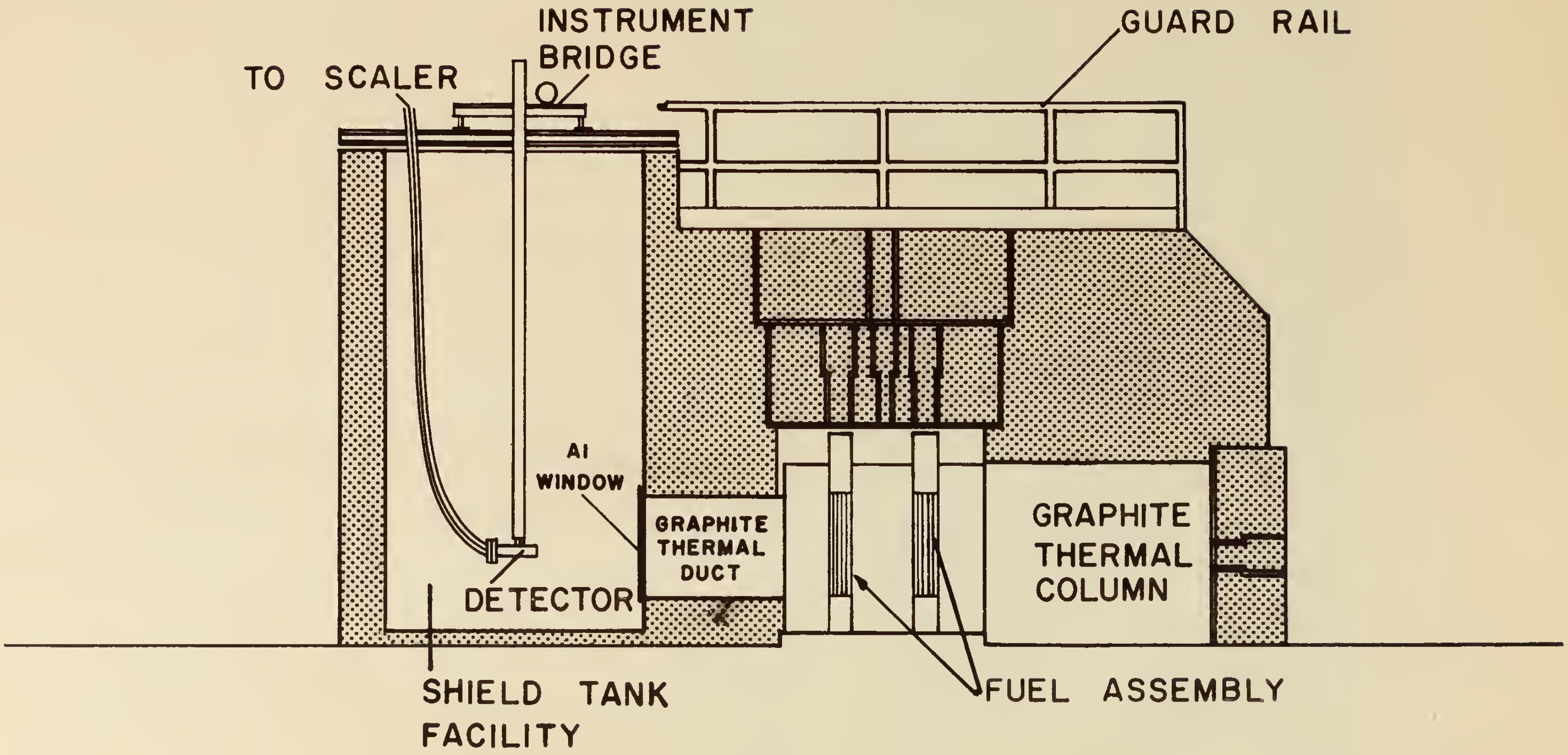
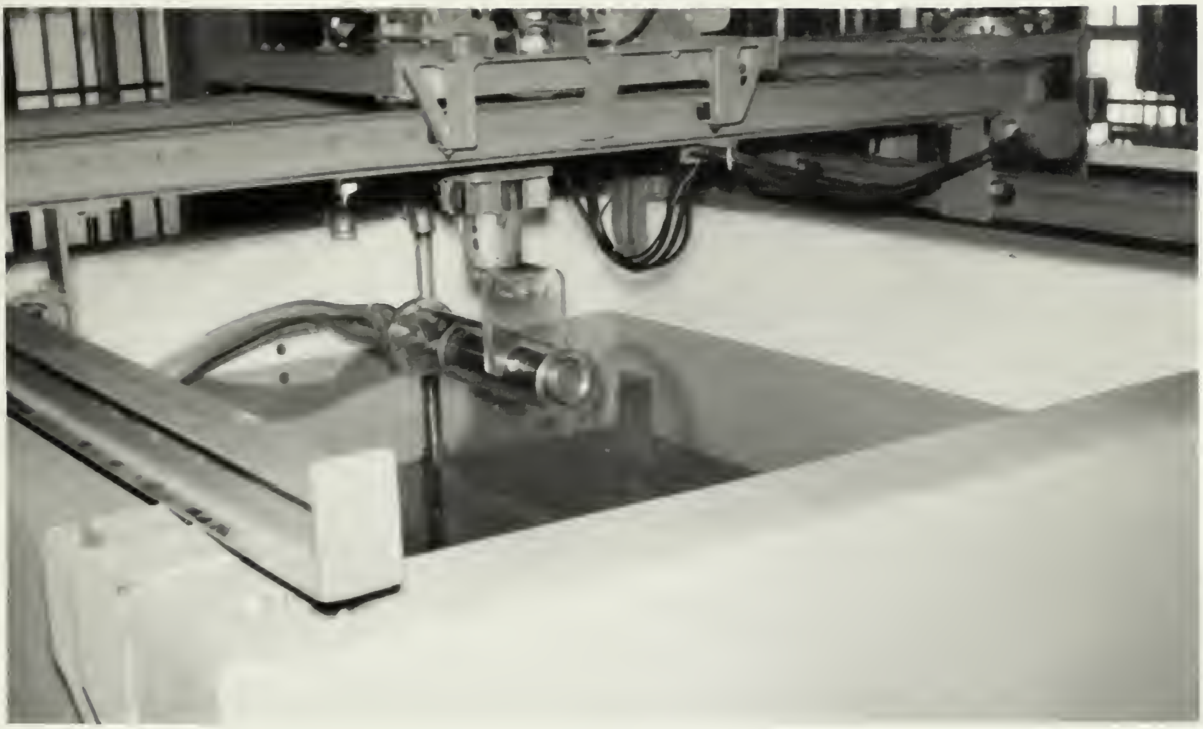

TO SCALER
INSTRUMENT BRIDGE
GUARD RAIL
Al WINDOW
GRAPHITE THERMAL DUCT
DETECTOR
GRAPHITE THERMAL COLUMN
SHIELD TANK FACILITY
FUEL ASSEMBLY

Figure 3. Scintillation detector attached to the instrument bridge



model DS5.

An aluminum rod and base device was used to hold a standardized Co^{60} source 40 cm from the detector. This arrangement was used to check the reproducibility in the electronic counting circuit. The standard run was made at a submerged depth of 100 cm to assure a low background.

The pulses from the scintillation detectors were registered on a Nuclear Chicago ultrascaler, model 192A-[illegible]. The resolution time for this scaler is one microsecond. The maximum counting rate is 1.7×10^5 cps.

The Nuclear Chicago radiation analyzer, model 1810, was used to investigate the gamma spectra. This instrument is only compatible with the DS5 scintillation detector.

The scaler and the detector were permitted to warm up for at least one hour prior to each run. Before the counts for the investigation were begun, a standard count was made. The background was counted at each of the 25 predesignated locations around the tank. The reactor was brought to the designated power, and the counts were not recorded until three consecutive counts at the first position showed no increase in the counting rate. This precaution was taken to assure that the reactor had reached steady state conditions prior to starting the counts. All counts corresponding to one power setting were made during the same run to eliminate inconsistences in matching reactor power with previous values. Complete runs were made at 0.1, 1.0, and 10 watts. Modified runs were made at 0.05 and 5.0 watts to verify some of the results obtained from the complete runs. During the modified runs, only a few positions in the tank were surveyed.

The radiation analyzer was permitted to warm up for a minimum of 15 hours prior to each run. Background was checked prior to each run and a

standard run was made to check reproducibility of the counting circuit. All readings corresponding to one power level were made during the same run. The radiation analyzer was standardized against a Cs^{137} source. The base level scale corresponded to energies from 0 to 2 Mev with a window width of 100 kev. All counts were made for 1 min.

V. RESULTS AND DISCUSSION

A. Centerline Gamma Measurements

Ten minute counts were made at 15 cm increments along the geometric centerline of the thermal duct window. These data are tabulated in Table 1 and plotted in Figures 4 and 5.

Figure 4 indicates the radiation level diminishes exponentially as the distance is increased between the window surface and the detector. This is in agreement with existing theories. The lines drawn in Figure 4 were fitted to the data points with a least squares fit. The mass absorption coefficients were determined to be 0.0444, 0.0444, 0.0423, 0.0409, and 0.0384 cm^2/gm for reactor power levels of 0.05, 0.1, 1.0, 5.0, and 10 watts, respectively. Grodstein (3) lists 0.0493 and 0.0396 cm^2/gm as the mass absorption coefficients for a collimated beam of monoenergetic x-rays with energies of 2 and 3 Mev, respectively.

From Figure 4, the linear approximation on the semi-log plot seems to be more satisfactory at the higher power levels. It is suggested that this is due to lessened effect of background and the better counting statistics at the higher counting rates. The background count may explain the higher readings at the 5 cm position on the 0.05 and 0.1 watt runs. Extrapolated values taken from Figure 4 indicate the counting rates at the window surface are 1.36 x 10^6, 5.30 x 10^6, and 9.30 x 10^6 cpm for power levels of 1.0, 5.0, and 10 watts, respectively.

Figure 5 is a log plot of the variation in count rate with reactor power for centerline positions 65, 80, 95, and 110 cm from the window. The general equation for the lines in this plot is

$$\text{count rate} = K\,(\text{reactor power})^n, \qquad (1)$$

Table 1. Experimental data for centerline intensities

Reactor Power (watts)	Distance from thermal duct window (cm)	Corrected count rate (cpm)	Standard Deviation (cpm)
0.05	5	77,829	279
0.05	20	34,180	151
0.05	35	16,812	130
0.05	50	8,340	91
0.05	65	4,345	66
0.05	80	2,242	49
0.05	95	1,240	35
0.05	110	720	27
0.1	5	153,831	392
0.1	20	71,020	266
0.1	35	34,418	185
0.1	50	17,310	131
0.1	65	8,862	94
0.1	80	4,608	68
0.1	95	2,646	51
0.1	110	1,416	38
1.0	35	342,041	585
1.0	50	175,276	418
1.0	65	92,182	306
1.0	80	47,684	218
1.0	95	26,321	162
1.0	110	15,035	123
5.0	50	716,007	846
5.0	65	387,695	623
5.0	80	204,976	452
5.0	95	115,000	339
5.0	110	64,001	253
10	65	775,136	881
10	80	425,722	653
10	95	239,048	489
10	110	137,700	371

Figure 4. Gamma intensity as a function of centerline displacement

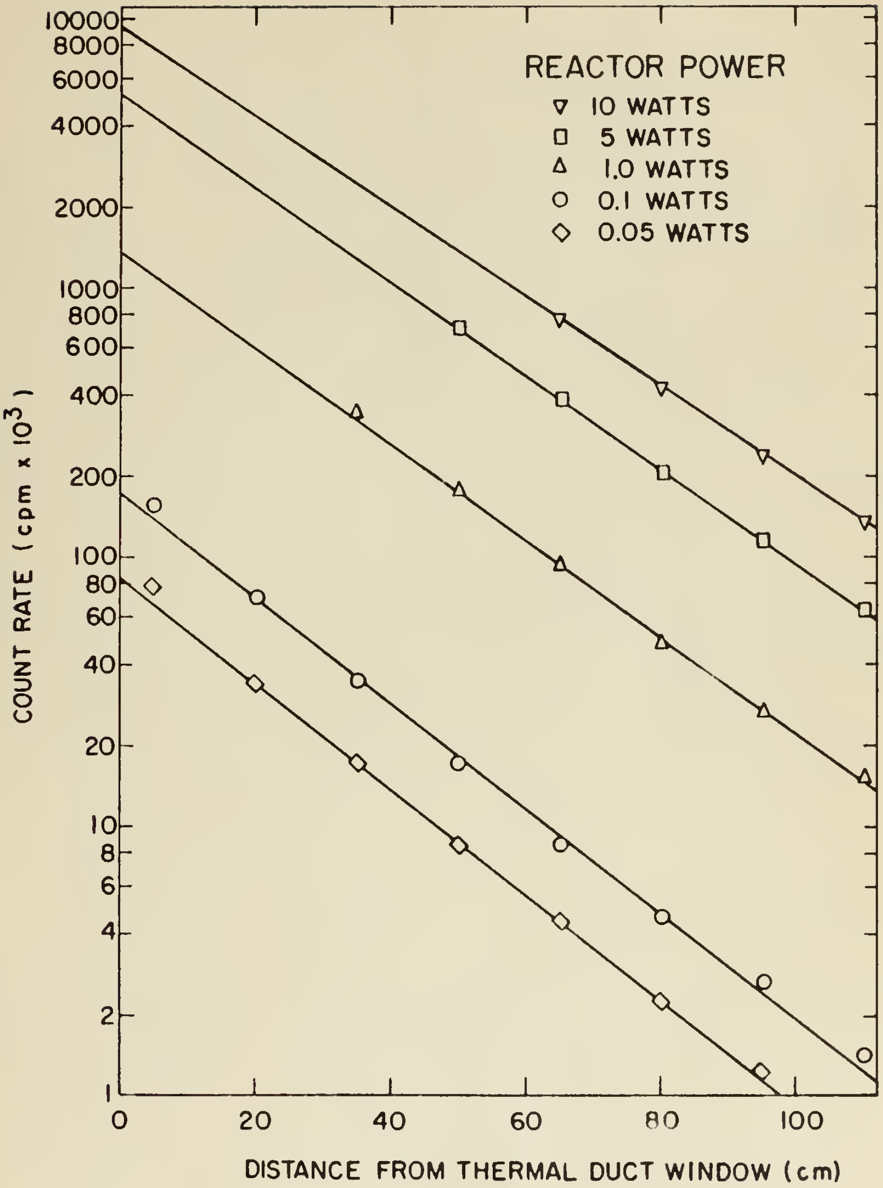
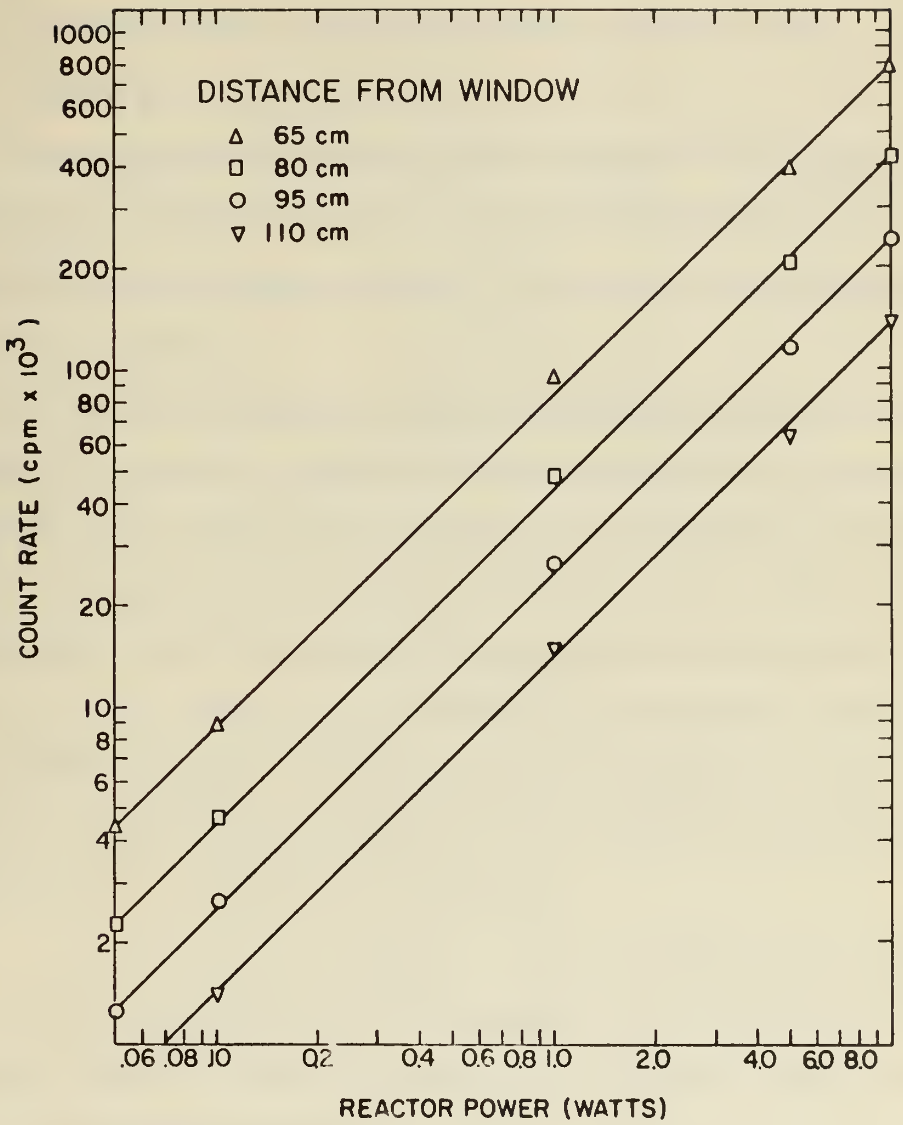

REACTOR POWER
▽ 10 WATTS
□ 5 WATTS
△ 1.0 WATTS
○ 0.1 WATTS
◇ 0.05 WATTS
COUNT RATE (cpm x 10^3)
10000
8000
6000
4000
2000
1000
800
600
400
200
100
80
60
40
20
10
8
6
4
2
1
0
20
40
60
80
100
DISTANCE FROM THERMAL DUCT WINDOW (cm)

Figure 5. Centerline gamma intensity as a function of reactor power


DISTANCE FROM WINDOW
△ 65 cm
□ 80 cm
○ 95 cm
▽ 110 cm
COUNT RATE (cpm x 10^3)
1000
800
600
400
200
100
80
60
40
20
10
8
6
4
2
.06
.08
.10
0.2
0.4
0.6
0.8
1.0
2.0
4.0
6.0
8.0
REACTOR POWER (WATTS)

The values of n were determined by a least squares fit to be 0.972, 0.979, 0.978, and 0.989 for the 65, 80, 95, and 110 cm positions, respectively. Since the values of n are nearly equal to one, the centerline counting rate is approximately a linear function of reactor power. The value of K is a function of position.

Only centerline positions having data for all five power levels tested were plotted in Figure 5. The same relationship seems to hold for all centerline positions, although only data for four positions are plotted in this figure.

The reactor instruments were the only method used to determine the operating power level. A check of the reproducibility of counting rates at the same power level on two different days, resulted in variations in the order of magnitude of the standard deviation of the count rate. If the run designated 1.0 watts was actually made at 1.05 watts, and the 5 watt run at 4.9 watts, this would aid in explaining why the actual deviation exceeded the standard deviation for the count rate for these two runs. It is felt that such variations are possible and may be due to small deviations from linearity of the reactor instrumentation.

F. Lateral and Vertical Variation of Gamma Intensity

In this portion of the investigation, it was desired to determine how the gamma radiation varied laterally and vertically across the window. It was also desired to examine the effect of reactor power on this distribution.

The data collected in the lateral direction are tabulated in Table 2 and plotted in Figure 6. All data were normalized against the centerline count rate for ease of plotting. It should be noted that these data were collected at 5, 50, and 95 cm from the window for 0.1, 1.0, and 10 watts, respectively. This permitted sufficiently high counting rates over the

entire lateral distance to provide the desired accuracy.

Examination of the data indicated that they were closely approximated by a cosine curve for the west side of the window but not on the east side. A study of the reactor construction details revealed that two control rods are located between the core and the tank, positioned symmetrically with respect to the core centerline. The rod on the west side is the number 1 safety rod, while the shim rod is located on the east side of the centerline. During normal operations the safety rod is completely removed from the core, while the shim rod was only displaced a small amount. This is believed to be the reason for the non-symmetry in these data. This is probably due to both the flux depression at the shim rod and the physical shielding effect of the rod itself. The curves drawn in Figure 6 are cosine curves represented by the general equation

$$y = \cos \pi x/2a. \qquad (2)$$

The values of a were determined to be 70, 83, and 93 cm for 0.1, 1.0, and 10 watts, respectively. These values were determined by a least squares fit applied to the data collected to the west of the centerline, which corresponds to the right side of Figure 6. The normalized count rate, corresponding to 57 cm east of the centerline, begins to approach the same cosine distribution, since the effects of the shim rod are diminished at this position.

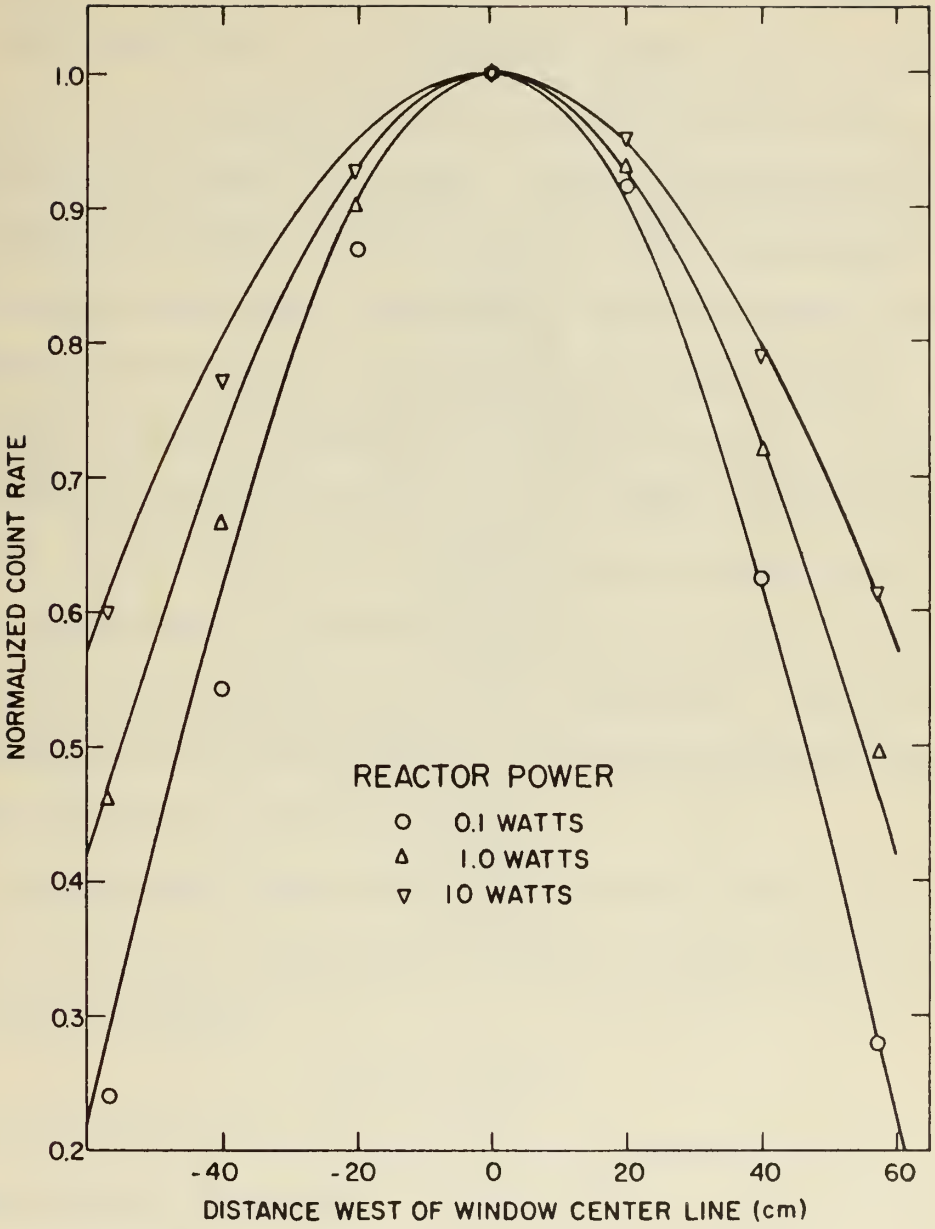
Even with the small deviation in the symmetry, it is felt that the lateral distribution of the gamma intensity is closely approximated by a cosine curve. It is suggested that this small deviation due to the shim rod will probably be less important than other experimental parameters in any system that may be investigated in this facility.

The data collected in the vertical direction across the face of the

Table 2. Experimental data for lateral intensity variation

Distance west of centerline (cm)	Normalized count rate (r_0 = ℄ count)	Standard deviation
Reactor power: 0.1 watts; distance from window: 5 cm; and r_0 = 157,765.		
57	0.27921	0.00151
40	0.62340	0.00253
20	0.91404	0.00333
0	1.00000	0.00356
- 20	0.86749	0.00321
- 40	0.54150	0.00230
- 57	0.34166	0.00138
Reactor power: 1.0 watts; distance from window: 50 cm; and r_0 = 173,054.		
57	0.49415	0.00207
40	0.71961	0.00258
20	0.93015	0.00322
0	1.00000	0.00316
- 20	0.90001	0.00314
- 40	0.66465	0.00253
- 57	0.46086	0.00197
Reactor power: 10 watts; distance from window: 95 cm; and r_0 = 239,840.		
57	0.61673	0.00204
40	0.79167	0.00245
20	0.95242	0.00270
0	1.00000	0.00289
- 20	0.92855	0.00273
- 40	0.77392	0.00239
- 57	0.60039	0.00200

Figure 6. Lateral distribution of gamma intensities


NORMALIZED COUNT RATE
1.0
0.9
0.8
0.7
0.6
0.5
0.4
0.3
0.2
-40
-20
0
20
40
60
DISTANCE WEST OF WINDOW CENTER LINE (cm)
REACTOR POWER
○ 0.1 WATTS
△ 1.0 WATTS
▽ 10 WATTS

window are tabulated in Table 3 and plotted in Figure 7. Again all data were normalized against the centerline count rate, and the distances from the window were 5, 50, and 95 cm for the three power levels.

The data points in Figure 7 show a definite lack of symmetry with respect to the centerline of the window; the count rate being markedly reduced near the bottom of the tank. The only non-symmetrical geometry in the vertical direction apparent from the reactor construction details is the concrete tank bottom. The effect of this non-symmetry was expected to increase as the distance between the window and the detector was increased. The data in Figure 7 indicate the inverse to be true. The largest deviation from symmetry occured in the 0.1 watt run, which was measured closest to the window. On this basis, it is suggested that the non-symmetry in the vertical distributions of the count rate is due to a lack of symmetry of the gamma field with respect to the window centerline. This may be due to either a displacement of the fuel assemblies or a non-symmetrical distribution of fission products in the fuel plates.

The data points above the centerline of the window may be approximated by a cosine curve. The curves in Figure 7 are cosine curves represented by the general equation

$$y = \cos \pi z/2b, \tag{3}$$

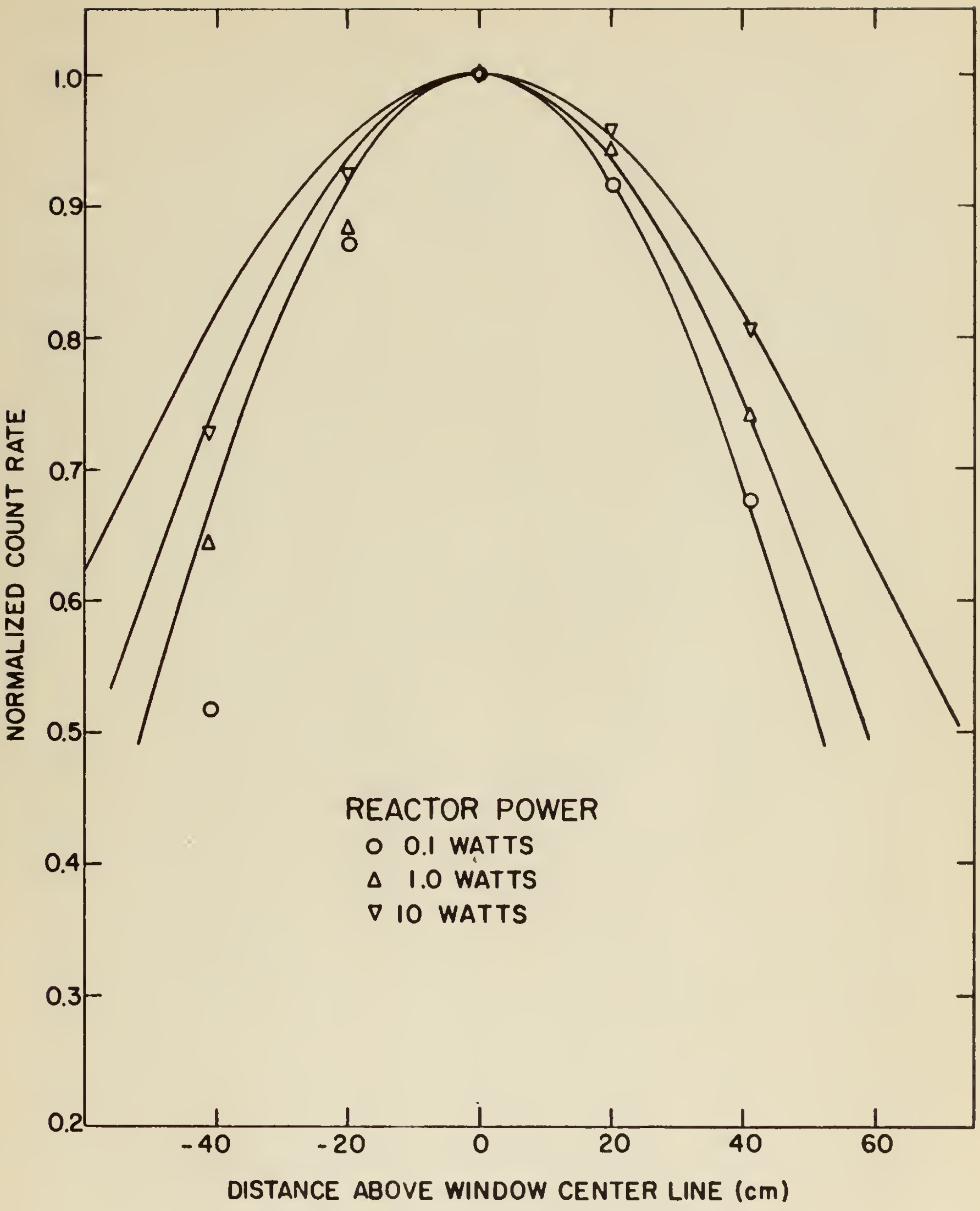
where the values of b were determined by a least squares fit to the data points above the centerline. On this basis, the values of b are 77, [illegible]7, and 103 for 0.1, 1.0, and 10 watts, respectively.

In future testing in this facility, a vertical cosine distribution of gamma intensity across the window would be highly questionable. If this assumption is made, it must be remembered that the intensities are markedly

Table 3. Experimental data for vertical intensity variation

Distance above centerline (cm)	Normalized count rate (r_o = ℄ count)	Standard deviation
Reactor power: 0.1 watts; distance from window: 5 cm; and r_o = 157,765.		
41	0.67614	0.00238
20	0.91539	0.00333
0	1.00000	0.00356
- 20	0.87313	0.00322
- 41	0.51605	0.00223
Reactor power: 1.0 watts; distance from window: 50 cm; and r_o = 173,064.		
41	0.73980	0.00272
20	0.94176	0.00326
0	1.00000	0.00340
- 20	0.88393	0.00310
- 41	0.64090	0.00247
Reactor power: 10 watts; distance from window: 95 cm; and r_o = 239,046.		
41	0.79671	0.00247
20	0.95917	0.00291
0	1.00000	0.00298
- 20	0.92400	0.00273
- 41	0.72961	0.00230

Figure 7. Vertical distribution of gamma intensities


1.0
0.9
0.8
0.7
0.6
0.5
0.4
0.3
0.2
NORMALIZED COUNT RATE
-40
-20
0
20
40
60
DISTANCE ABOVE WINDOW CENTER LINE (cm)
REACTOR POWER
○ 0.1 WATTS
△ 1.0 WATTS
▽ 10 WATTS

reduced near the bottom of the tank.

C. Gamma Spectral Analysis

The radiation analyzer was used to determine the variation in the gamma spectrum with reactor power at a centerline position 55 cm from the thermal duct window. The spectrum was analyzed between 0 and 2.0 Mev with a window width 100 kev. The decision to investigate this portion of the spectrum was based on the fact that less than ten per cent of the total gammas counted were more energetic than 2 Mev. The scale on the analyzer was calibrated prior to each run with a C_s^{137} source. It is known that the error in spectral energy measurements increases away from the calibration point, thus making the absolute values of the energy scale questionable at the end points.

The data tabulated in Table 4 corresponds to reactor powers of 0.1, 1.0, and 10 watts. These data were normalized by the counting rate of the lowest energy interval and plotted in Figure 8. This figure indicates close agreement for all three power levels. It is noted that although the data points for all three power settings are nearly equal, there is a tendency for the counting rates corresponding to the higher power levels to be slightly increased. There is a complete lack of characteristic peaks in these data.

Spectral data were also collected for two other positions in the tank; on the centerline, 5 cm from the window at a power of 0.1 watts; and 57 cm east of the centerline, 65 cm from the window at 10 watts. These data are given in Table 5 and are plotted in Figure 8.

The data corresponding to the 5 cm-0.1 watt run does show variation from the other spectral data plotted, although the general shape of the

Table 4. Experimental spectral data for power variation

(On centerline; distance from the window: 65 cm)

Gamma Energy (Mev)	Normalized count rate	Standard deviation
Reactor power: 0.1 watts; normalizing count rate: 5,259 cpm.		
0.050	1.00000	0.01970
0.150	0.44052	0.01109
0.250	0.10174	0.00908
0.350	0.06040	0.00390
0.450	0.04030	0.00283
0.550	0.02758	0.00231
0.650	0.01944	0.00189
0.750	0.01407	0.00167
0.850	0.01122	0.00146
0.950	0.00666	0.00113
1.050	0.00875	0.00130
1.150	0.00773	0.00127
1.250	0.00875	0.00130
1.350	0.00666	0.00113
1.450	0.00551	0.00103
1.550	0.00627	0.00109
1.650	0.00455	0.00093
1.750	0.00513	0.00099
1.850	0.00361	0.00083
1.950	0.00546	0.00111
2.050	0.00323	0.00078

Table 4. (continued)

(On centerline; distance from the window: 65 cm)

Gamma Energy (Mev)	Normalized count rate	Standard deviation
Reactor power: 1.0 watts; normalizing count rate: 52,474 cpm.		
0.050	1.00000	0.00617
0.150	0.49240	0.00273
0.250	0.17700	0.00200
0.350	0.07465	0.00126
0.450	0.04593	0.00096
0.550	0.02977	0.00076
0.650	0.02048	0.00063
0.750	0.01610	0.00056
0.850	0.01297	0.00050
0.950	0.01081	0.00048
1.050	0.01017	0.00044
1.150	0.00904	0.00042
1.250	0.00826	0.00040
1.350	0.00731	0.00038
1.450	0.00603	0.00034
1.550	0.00597	0.00034
1.650	0.00578	0.00033
1.750	0.00610	0.00035
1.850	0.00480	0.00031
1.950	0.00458	0.00030
2.050	0.00401	0.00031

Table 4. (continued)

(On centerline; distance from the window: 65 cm)

Gamma Energy (Mev)	Normalized count rate	Standard deviation
Reactor power: 10 watts; normalizing count rate: 464,428 cpm.		
0.050	1.00000	0.00200
0.150	0.40440	0.00137
0.250	0.13156	0.00068
0.350	0.07265	0.00044
0.450	0.04136	0.00033
0.550	0.03053	0.00025
0.650	0.02174	0.00021
0.750	0.01764	0.00020
0.850	0.01477	0.00018
0.950	0.01273	0.00017
1.050	0.01108	0.00016
1.150	0.00952	0.00015
1.250	0.00910	0.00014
1.350	0.00802	0.00013
1.450	0.00739	0.00013
1.550	0.00661	0.00012
1.650	0.00633	0.00012
1.750	0.00562	0.00011
1.850	0.00543	0.00011
1.950	0.00491	0.00010
2.050	0.00453	0.00010

Table 5. Experimental spectral data for position variation

(On centerline; distance from the window: 5 cm)

Gamma Energy (Mev)	Normalized count rate	Standard deviation
Reactor power: 0.1 watts; normalizing count rate: 69,332 cps.		
0.050	1.00000	0.00537
0.150	0.67054	0.00399
0.250	0.26304	0.00209
0.350	0.11940	0.00153
0.450	0.08417	0.00115
0.550	0.06013	0.00096
0.650	0.04298	0.00080
0.750	0.03809	0.00075
0.850	0.03452	0.00072
0.950	0.03144	0.00068
1.050	0.02943	0.00065
1.150	0.02960	0.00066
1.250	0.02713	0.00063
1.350	0.02367	0.00059
1.450	0.02260	0.00057
1.550	0.01974	0.00054
1.650	0.01863	0.00052
1.750	0.01767	0.00052
1.850	0.01572	0.00048
1.950	0.01451	0.00046
2.050	0.01125	0.00041

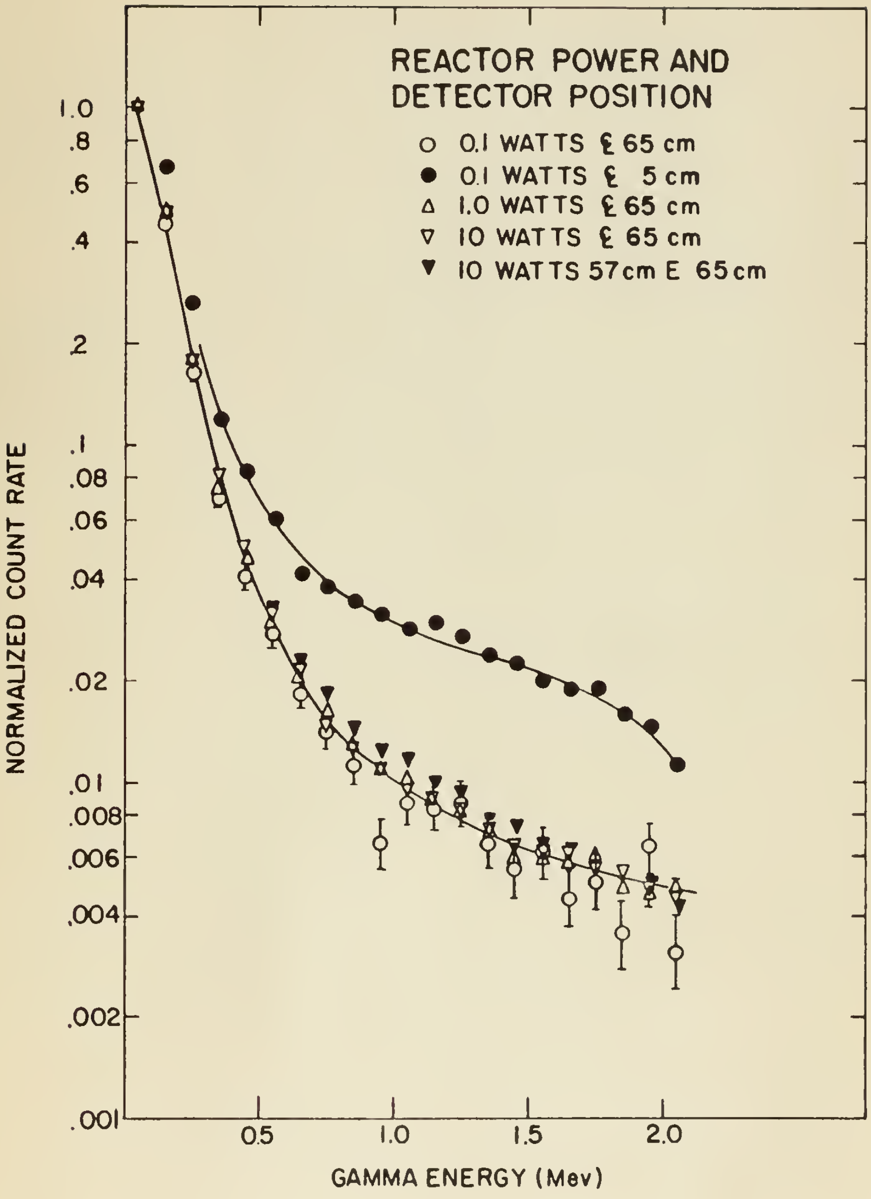
Table 5. (continued)

(57 cm east of centerline; 65 cm from window)

Reactor power: 10 watts; normalizing count rate: 230,053 cpm.

Gamma Energy (Mev)	Normalized count rate	Standard deviation
0.050	1.00000	0.00265
0.150	0.31717	0.00105
0.250	0.10906	0.00079
0.350	0.05107	0.00062
0.450	0.04678	0.00046
0.550	0.03350	0.00040
0.650	0.02325	0.00032
0.750	0.01839	0.00025
0.850	0.01475	0.00025
0.950	0.01248	0.00024
1.050	0.01192	0.00024
1.150	0.01003	0.00023
1.250	0.00856	0.00022
1.350	0.00795	0.00020
1.450	0.00758	0.00018
1.550	0.00670	0.00017
1.650	0.00642	0.00015
1.750	0.00585	0.00015
1.850	0.00529	0.00015
1.950	0.00510	0.00014
2.050	0.00442	0.00014

Figure 8. Gamma spectrum variation with reactor power and tank position


REACTOR POWER AND DETECTOR POSITION
○ 0.1 WATTS ℄ 65 cm
● 0.1 WATTS ℄ 5 cm
△ 1.0 WATTS ℄ 65 cm
▽ 10 WATTS ℄ 65 cm
▼ 10 WATTS 57cm E 65cm
NORMALIZED COUNT RATE
1.0
.8
.6
.4
.2
.1
.08
.06
.04
.02
.01
.008
.006
.004
.002
.001
0.5
1.0
1.5
2.0
GAMMA ENERGY (Mev)

curve remained nearly the same. It was noted that although the counting rate of the 10 watt run had been diminished by a factor of 2 by displacing the chamber 57 cm from the centerline, the normalized data points were nearly identical. It would appear that the change in the 5 cm-0.1 watt spectrum was due to the effect of 60 cm less water thickness. The displacement from the centerline did not appreciably affect the thickness of the water seen by the gammas in the 10 watt run.

From Figure 8 it appears that the gamma spectrum in the tank is nearly independent of the power level of the reactor. This spectrum will vary slightly, depending on the distance from the window, due to the water thickness seen by the gammas.

VI. CONCLUSIONS

As a result of this investigation, the following has been concluded:

1. The centerline gamma intensity is approximately a linear function of reactor power.

2. The gamma intensity varies laterally across the thermal duct window as a cosine function. Intensities are slightly reduced to the east of the centerline due to the presence of the shim rod.

3. The gamma intensity varies vertically across the thermal duct window as a cosine function only above the centerline. The intensities are reduced markedly near the bottom of the tank.

4. The gamma spectrum in the tank is nearly independent of the power level, but does vary somewhat with the thickness of the water seen by the radiation.

VII. LITERATURE CITED

1. Cady, O.P. The Lid Tank Shielding Facility at Oak Ridge National Laboratory, Part III. Instrumentation. U.S. Atomic Energy Commission Report ORNL-2587. (Oak Ridge National Lab., Tenn.). 1960.

2. Casper, A.W. Comparison of bulk shielding reactor centerline measurements in water with predictions. U.S. Atomic Energy Commission Report APEX-504. (General Electric Co. Aircraft Nuclear Propulsion Dept., Cincinnati). 1959.

3. Grodstein, G.W. X-ray attenuation coefficients from 10 kev to 100 Mev. U.S. National Bureau of Standards Circular 583. 1957.

4. Maienschein, F.C., Peele, R.W., and Love, T.A. Recent fission gamma-ray spectral measurements. U.S. Atomic Energy Commission Report TID-6302. (Technical Information Service Extension, AEC). 1961.

5. Morgan, W.L., Epstein, H.M., Anno, J.N., Jr., and Chastain, J.W., Jr. Shielding-research area at Battelle. U.S. Atomic Energy Commission Report BMI-1291. (Battelle Memorial Inst., Columbus, Ohio). 1958.

6. Price, W.J. Nuclear radiation detection. New York, N.Y., McGraw-Hill Book Co., Inc. 1958.

7. Rafferty, D.J. Thermal neutron shielding measurements in the UTR-10 shield tank facility. Unpublished M.S. Thesis. Ames, Iowa, Library, Iowa State University of Science and Technology. 1960.

VIII. ACKNOWLEDGMENTS

The author wishes to thank Dr. Glenn Murphy for his suggestions and helpful assistance throughout this project. Appreciation is also extended to Mrs. E.L. Denison and Mr. R.A. Danofsky who operated the reactor for the experimental portion of this project.

Finally, I would like to express my appreciation to the United States Navy for the opportunity to obtain an advanced technical education at the U. S. Naval Postgraduate School and Iowa State University.

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